NEUTRONIC FEASIBILITY STUDIES USING U-Mo DISPERSION FUEL (9 Wt % Mo, 5.0 gU/cm^3) FOR LEU CONVERSION OF THE MARIA (POLAND), IR-8 (RUSSIA), AND WWR-SM (UZBEKISTAN) RESEARCH REACTORS*

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Presented at the
2000 International Meeting on Reduced Enrichment
for Research and Test Reactors

October 1-6, 2000
Las Vegas, Nevada USA

*Work supported by the U.S. Department of Energy
Office of Nonproliferation and National Security
under Contract No. W-31-109-ENG-38

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ABSTRACT

U-Mo alloys dispersed in an Al matrix offer the potential for high-density uranium fuels needed for the LEU conversion of many research reactors. On-going fuel qualification tests by the US RERTR Program show good irradiation properties of U-Mo alloy dispersion fuel containing 7-10 weight percent molybdenum. For the neutronic studies in this paper the alloy was assumed to contain 9 wt % Mo (U-9Mo) with a uranium density in the fuel meat of 5.00 gU/cm^3 which corresponds to 32.5 volume % U-9Mo. Fuels containing U-9Mo have been used in Russian reactors since the 1950's. For the three research reactors analyzed here, LEU fuel element thicknesses are the same as those for the Russian-fabricated HEU reference fuel elements.

Relative to the reference fuels containing 80-90% enriched uranium, LEU U-9Mo Al-dispersion fuel with 5.00 gU/cm^3 doubles the cycle length of the MARIA reactor and increases the IR-8 cycle length by about 11%. For the WWR-SM reactor, the cycle length, and thus the number of fuel assemblies used per year, is nearly unchanged. To match the cycle length of the 36% enriched fuel currently used in the WWR-SM reactor will require a uranium density in the LEU U-9Mo Al-dispersion fuel of about 5.4 gU/cm^3. The 5.00 gU/cm^3 LEU fuel causes thermal neutron fluxes in water holes near the edge of the core to decrease by (6-8)% for all three reactors.

INTRODUCTION

The irradiation behavior of small-sample U-Mo alloy dispersion fuels at low temperature is summarized in Ref.'s 1-2. These RERTR tests show good irradiation behavior for Mo weight fractions of 7-10% and burn-ups to 70%. Combining these results with existing experience in fabricating high-volume-loaded dispersion fuels increases the probability that for this type fuel uranium densities of 8-9 g/cm^3 will become available for rolled fuel plates and 5-6 gU/cm^3 for extruded fuel tubes. The Russian RERTR program reported[3] fabrication experience with U-9Mo alloy dispersed in Al at a volume fraction of 40% (≈6.1 gU/cm^3) and a hot-pressing temperature of 450-500^{0}C. Plans for irradiation-tests of this fuel in 2 extruded fuel elements of the IVV-2M type were also reported. Once qualified, these high-density fuels will make possible the LEU conversion of most research reactors.

This study examines the feasibility for LEU conversion of three Russian-designed research reactors using U-9Mo Al-dispersion fuel with a density of 5.0 gU/cm^3. The MARIA reactor (Swierk, Poland), the IR-8 reactor (Moscow, Russia) and the WWR-SM reactor (Ulugbek, Uzbekistan) are described in Ref.'s 4-6. These papers examined the feasibility of using UO_2-Al dispersion fuel for the LEU conversion of these reactors. Conversion of the IR-8 and WWR-SM reactors required the use of IRT-4M fuel assemblies that have thicker meat, thicker fuel elements, and thinner coolant channels than the IRT-3M assemblies in current use. UO_2 volume fractions greater than 40% were needed, but in this range (3.85 gU/cm^3, 43% volume fraction) failures of Russian extruded fuel assemblies have been reported[7]. For tubular fuel elements used in many Russian-designed research reactors, the practical fabrication limit for UO_2-Al fuel appears to be about 36% by volume (~3.2 gU/cm^3) [8]. LEU conversions with U-9Mo Al-dispersion fuel (5.0 gU/cm^3) allows the use of lower volume fractions while maintaining the IRT-3M fuel assembly geometry. Control rod worths were not evaluated in this study. They are expected to be nearly the same as those found previously for LEU UO_2-Al fuel.

FUEL ASSEMBLIES AND REACTOR CORE CONFIGURATIONS

The cross section of Russian-fabricated M6-type 6-tube fuel assemblies used in the MARIA research reactor is shown in Fig. 1. Fuel assemblies are surrounded by beryllium and are located on a square grid with a 13.0-cm pitch at the core midplane. Figure 2 shows the 16-fuel assembly configuration for the 17 MW reference core used for these HEU-to-LEU conversion studies. Table 1 provides geometry and uranium loading parameters for the MARIA reactor M6-type fuel assemblies. Ref. 4 gives the results of earlier studies based on the use of LEU UO_2-Al dispersion fuel. This reference also describes how estimates were made for ^{6}Li and ^{3}He poison concentrations for in-core and ex-core beryllium blocks. These same poison concentrations were used in these studies based on U-9Mo Al-dispersion fuel.

Table 1. M6 Fuel Assembly Parameters
(For the MARIA Research Reactor)

^{235}U enrichment, wt %	80.0	19.7	19.7
Fuel meat	U-Al Alloy	UO_2-Al	U-9Mo Al[a]
Dispersant density, g/cm^3	6.34	~10 [b]	16.9
Dispersant volume fraction, %	28.3	28.7	32.5
Uranium weight fraction, %	71.3	88.1	91.0
Uranium density in meat, gU/cm^3	1.28	2.53	5.00
Meat/clad/element thickness, mm	.40/.80/2.00	.94/.53/2.00	.60/.70/2.00
Coolant channel thickness, mm	2.50	2.50	2.50
Meat length, mm	1000	1000	1000
^{235}U mass per fuel assembly, g	350	402	506

[a] U-9Mo is a U-Mo alloy containing 9 wt % Mo.

[b] The effective density of UO_2 was assumed to be ~10 g/cm^3 which is about 90% of the theoretical density of the UO_2 – U_4O_9 mixed phase system.

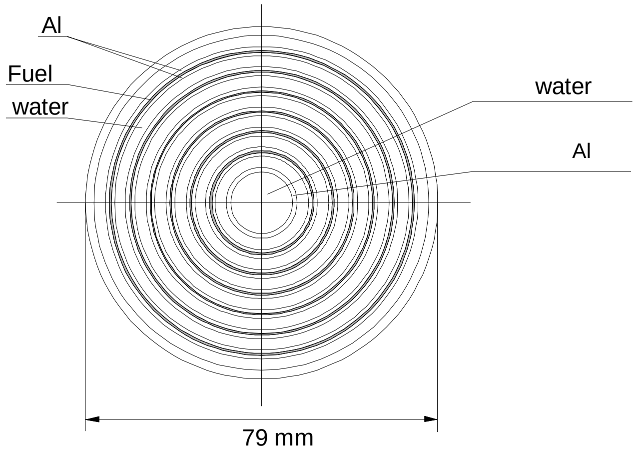
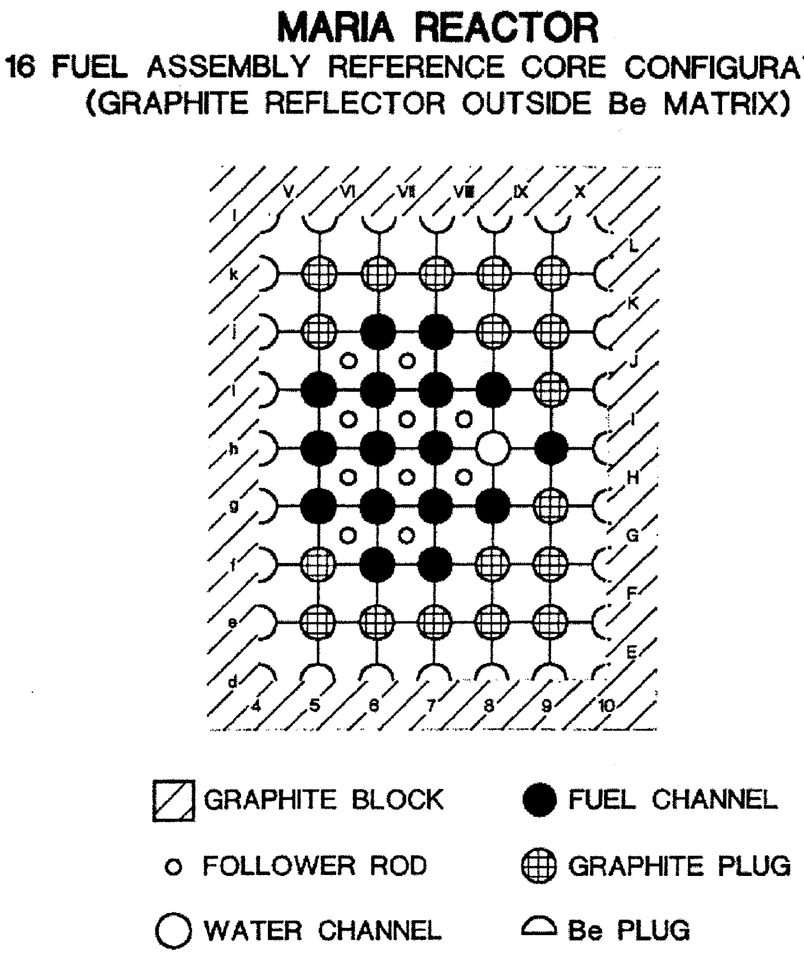


Figure 1. Horizontal Cross Section of the MARIA Reactor M6 Fuel Assembly



Location of Safety Rods: H-VI, J-VI, G-VII, I-VII, H-VIII

Figure 2 . Location of Control Rods: G-VI, I-VI, J-VII, I-VIII, H-VII

Figure 3 shows a cross section of the Russian-designed 6-tube and 8-tube IRT-3M and IRT-4M fuel assemblies. IRT-3M fuel is used in both the IR-8 and the WWR-SM research reactors. The IRT-4M design was used in the earlier LEU conversion studies with UO_2-Al dispersion fuel reported in Ref.'s 5 and 6. To maintain the same water channel thickness as the HEU reference fuel, these LEU U-9Mo Al-dispersion conversion studies used IRT-3M fuel assemblies with the same fuel meat, clad, and element thicknesses as the tubes with 36% enriched uranium. Table 2 gives the geometry and uranium loading parameters for the IRT-3M assemblies.

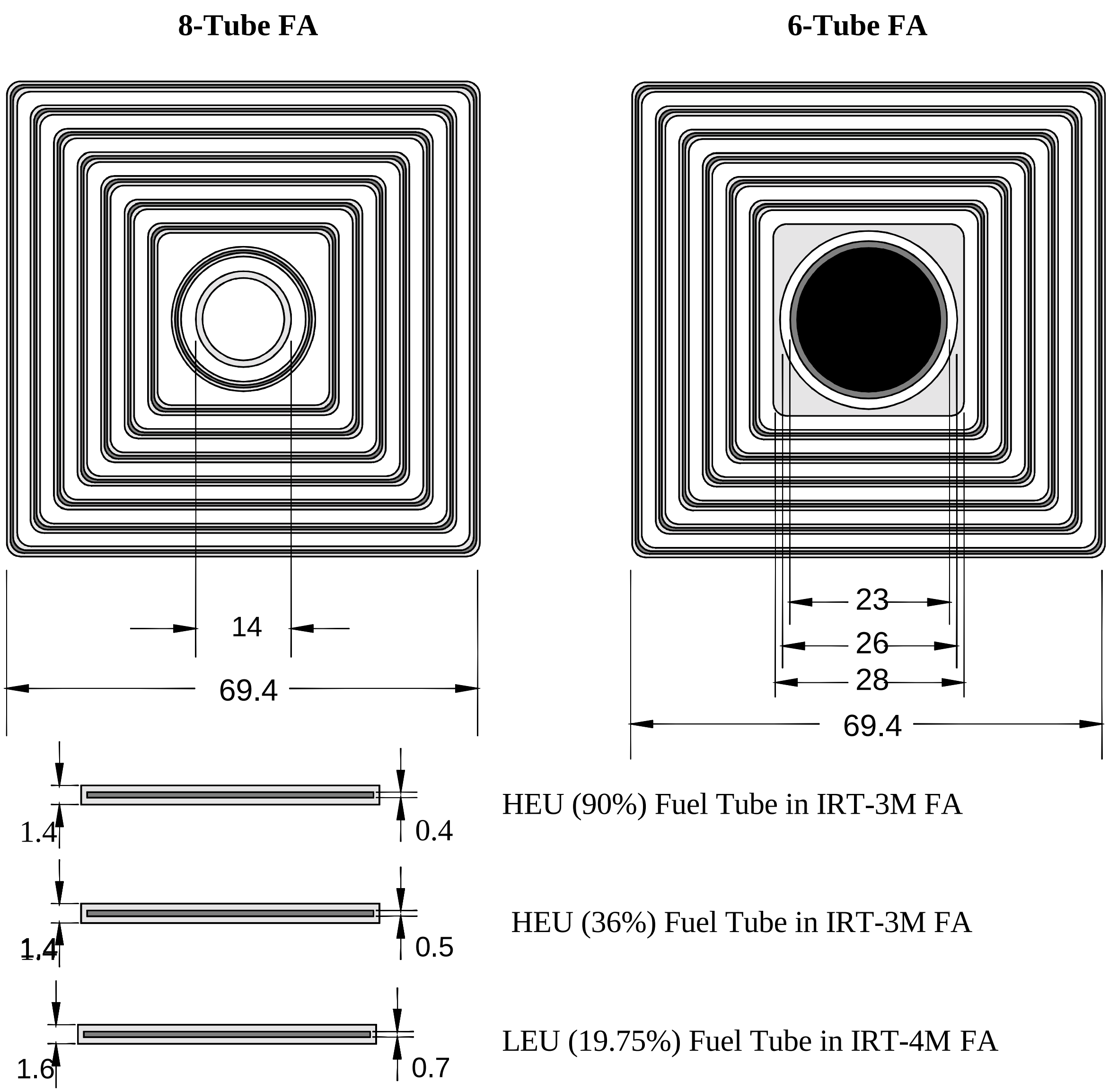


Figure 3. Horizontal Cross Section of the IRT-3M and IRT-4M Fuel Assemblies

Table 2. IRT-3M Fuel Assembly Parameters
(For the IR-8 and WWR-SM Research Reactors)

^{235}U enrichment, wt %	**90.0**	**36.0**	**19.75**
Fuel meat	**UO_2-Al**	**UO_2-Al**	**U-9Mo Al [a]**
Dispersant (UO_2 or U-9Mo) density-g/cm^3	**~10 [b]**	**~10 [b]**	**16.9**
Volume fraction (UO_2 or U-9Mo), %	**12.1**	**28.5**	**32.5**
Uranium weight fraction, %	**88.1**	**88.1**	**91.0**
Uranium density in meat, gU/cm^3	**1.07**	**2.51**	**5.00**
Meat/clad/element thickness, mm	**.40/.50/1.40**	**.50/.45/1.40**	**.50/.45/1.40**
Coolant channel thickness, mm	**2.05**	**2.05**	**2.05**
Meat length, mm	**580**	**580**	**580**
^{235}U mass per fuel assembly, 6/8 tube, g	**264/300**	**309/351**	**339/385**

[a] U-9Mo is a U-Mo alloy containing 9 wt % Mo.
[b] The effective density of UO_2 was assumed to be ~10 g/cm^3.

Figure 4 shows the core configuration for the 8 MW, beryllium-reflected, IR-8 reactor[5]. Arranged in a 4x4 array, the active core consists of 16 IRT-3M 6-tube fuel assemblies. Reactivity effects of ^{6}Li and ^{3}He poisons in the beryllium reflector were neglected in the calculations because no information is available from which the concentration of these poisons can be estimated.

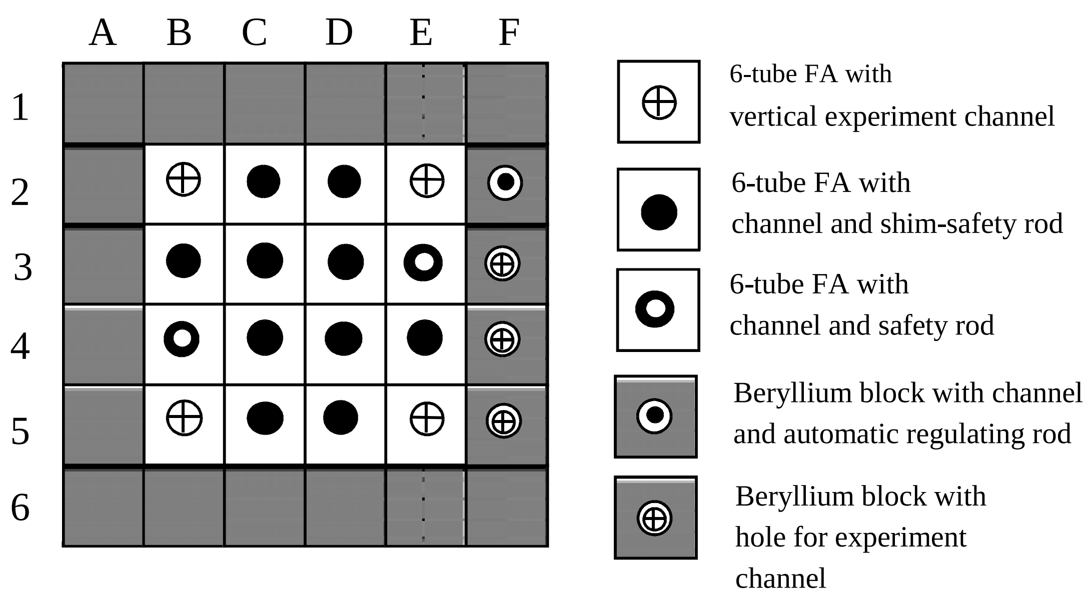


Figure 4. Core Configuration for the IR-8 Reactor

Figure 5 shows the core configuration for the WWR-SM reactor. It consists of 20 IRT-3M 6-tube fuel assemblies and 4 IRT-3M 8-tube fuel assemblies located in outer core grid positions. The core is beryllium-reflected and also has beryllium blocks that occupy the central 4 grid positions. Rough estimates for ^{3}He and ^{6}Li poison levels in the inner and outer beryllium blocks are given in Ref. 5. These same concentrations were used in the current LEU conversion studies with U-9Mo Al-dispersion fuel.

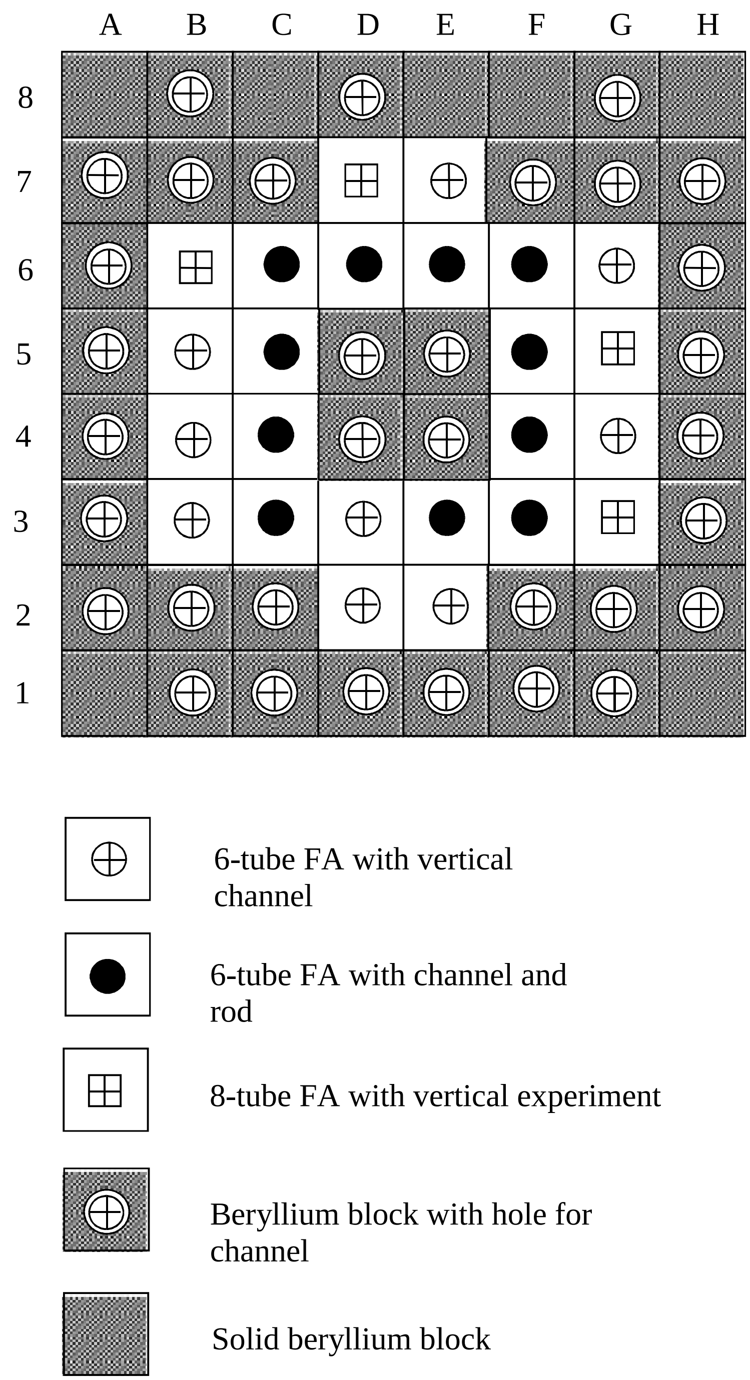


Figure 5. Core Configuration for the WWR-SM Reactor

NEUTRONIC CALCULATIONS

Multigroup cross sections were generated using the WIMS-ANL code[9] and its 69-group library based on ENDF/B-VI data. The cross sections were collapsed into 7 broad groups with energy boundaries of 1.0E+7, 8.21E+5, 5.53E+3, 4.0E+0, 6.25E-1, 2.5E-1, 5.8E-2 and 1.0E-5 eV. These cross sections were used with the REBUS-3 fuel cycle analysis code[10]. DIF3D[11]-diffusion and MCNP[12]-Monte Carlo calculations were performed for each core configuration with fresh fuel to determine DIF3D/MCNP reactivity bias factors. Beam tubes were included in the MCNP calculations but not in the DIF3D models. These bias factors were applied to the excess reactivities calculated by REBUS-3 at the beginning and end of the equilibrium cycles.

Equilibrium fuel cycle analyses for the MARIA reactor are based on a fuel shuffling scheme in which one fresh fuel assembly is loaded at a central core location and one spent fuel assembly is removed from the edge of the core at the beginning of the equilibrium cycle (BOEC). Thus, each fuel assembly remains in the core for 16 burn cycles. An in/out fuel management strategy was also used for the IR-8 reactor calculations. For this case, however, two fresh 6-tube fuel assemblies were loaded at the beginning of each cycle and two were discharged after remaining in the core for 8 burn cycles. For the WWR-SM reactor, two fresh 6-tube IRT-3M fuel assemblies are loaded near the center of the core and are gradually moved outward until they are discharged from the edge of the core after 10 burn cycles. One 8-tube IRT-3M fuel assembly is discharged after 24 burn cycles. Ref.'s 4-6 provide more details concerning the fuel shuffling schemes used for each of these three reactors.

All REBUS-3 calculations were done in XYZ geometry. For the IR-8 and WWR-SM models it was assumed that the core is symmetric about the core midplane. The central hole in the 6-tube IRT-3M fuel assemblies (see Fig. 3) was filled with either water or the aluminum alloy control rod followers. For the MARIA reactor calculations midplane symmetry was not assumed and all control rods were fully withdrawn into the upper reflector. Each IRT-3M fuel assembly was divided into four axial depletion zones. The MARIA reactor M6 fuel assemblies were divided into 5 axial depletion zones.

RESULTS FROM EQUILIBRIUM FUEL CYCLE ANALYSES

MARIA Reactor:

Table 3 summarizes results obtained from REBUS-3 equilibrium fuel-cycle calculations for LEU U-9Mo Al-dispersion (5.00 gU/cm^3) M6-type fuel assemblies (see Table 1 for meat, clad and coolant thicknesses). Results for the HEU (80%) reference-fuel are taken from Ref. 4. The MARIA reactor currently uses this HEU fuel and has a licensing requirement that limits the ^{235}U average discharge burnup to 45%. Use of U-9Mo Al-dispersion fuel and 45% discharge burnup would increase the cycle length from 7.5 to 11.2 full power days. Put another way, the U-9Mo fuel would reduce the number of fuel assemblies consumed per year from 21 to 14 based on the 1997 value of 3856 hours on power per year[4]. However, U-9Mo Al-dispersion fuel assemblies could operate safely for discharge burn-ups of at least 70% if irradiation testing is successful.

The BOEC and EOEC excess reactivities shown in Table 3 have been corrected for the DIF3D/MCNP reactivity bias which was calculated for fresh fuel. The Monte Carlo calculations include beam tubes in the reactor model. However, they are omitted in the model used for calculations based on diffusion theory. Use of LEU U-9Mo Al dispersion fuel with 5.00 gU/cm^3 would double the cycle length of the HEU reference fuel for the same EOEC excess reactivity. For this case the number of fuel assemblies used per year would be reduced from 21.4 to 10.5. The peak thermal neutron flux in location h8 (see Fig.2) is reduced by about 7%. Some results from Table 3 are plotted in Fig. 6.

Table 3. MARIA Research Reactor Equilibrium Fuel Cycle Comparisons (Core: 16 M6 6-Tube Fuel Assemblies, Power = 17.0 MW)

Calc. Type	^{235}U Enr. wt%	Fuel Meat (^{235}U-g/FA)	Cycle length fpd's	Excess Reactivity, % δk/k [a]				Discharge BU		Peak Flux[b]	FA's used per yr[c]
				BOL	Bias	BOEC	EOEC	Ave. %	Peak %	Loc. h8	
MCNP	80.0	U-Al Alloy	0.0	16.26	0.0						
DIF3D	80.0	(350)	0.0	18.39	2.13						
Rebus3	80.0	(350)	7.51		2.13	4.61	2.88	45.0	56.1	2.73	21.4
		(350)	8.50		2.13	3.17	1.13	50.9	62.5	2.70	18.9
Rebus3	19.7	UO_2-Al	8.72		2.13	4.98	3.42	44.3	55.2	2.66	18.4
		(402)									
MCNP	19.7	U-9Mo Al	0.0	15.75	0.0						
DIF3D	19.7	(506)	0.0	16.51	0.75						
Rebus3	19.7	(506)	11.16		0.75	8.67	7.17	45.0	56.1	2.61	14.4
Rebus3	19.7	(506)	12.38		0.75	7.74	6.04	49.7	61.2	2.59	13.0
Rebus3	19.7	(506)	13.65		0.75	6.69	4.74	54.5	66.2	2.56	11.8
Rebus3	19.7	(506)	14.86		0.75	5.59	3.38	58.9	70.8	2.54	10.8
Rebus3	19.7	(506)	16.10		0.75	4.35	1.83	63.5	75.2	2.52	10.0

[a] Excess reactivities at the beginning of the equilibrium cycle (BOEC) and at the end of the equilibrium cycle (EOEC) have been corrected for the DIF3D/MCNP reactivity bias.
[b] These thermal neutron fluxes (E_n < 0.625 eV) are in units of E+14 n/cm^2–sec and are multiplied by k_{eff} to adjust them to the critical core condition.
[c] Fuel consumption estimates are based on the 1997 value of 3856 hours on power per year[4].

Special calculations were done for the MARIA reactor to study the effects of increasing both clad thickness and uranium density for a fixed fuel tube thickness of 2.00 mm and an average ^{235}U burnup of 45%. The results of this study are summarized in Table 4 and are plotted in Fig. 7. For the same clad thickness as is used for the HEU fuel assemblies, a uranium loading of about 6.0 g/cm^3 in the LEU U-9Mo Al-dispersion fuel would be required.

Table 4. MARIA Research Reactor Equilibrium Fuel Cycle Calculations Cycle Length = 9.22 fpd's, Average ^{235}U Discharge Burnup = 45.0% (Fuel: LEU (19.7%), U-9Mo Al, 418g ^{235}U per FA)

Wt % ^{235}U	Fuel Meat	t_{clad}[a] mm	t_{meat} mm	Density gU/cm^3	Vol Fract, %	Excess Reactivity % δk/k	
						BOEC	EOEC
19.7	U-9Mo Al	0.53	0.94	2.63	17.1	5.90	4.24
"	"	0.69	0.62	4.00	26.0	5.78	4.20
"	"	0.75	0.50	5.00	32.5	5.72	4.14
"	"	0.79	0.42	6.00	39.0	5.61	4.03

[a] The clad thickness for the HEU reference fuel is 0.80 mm.

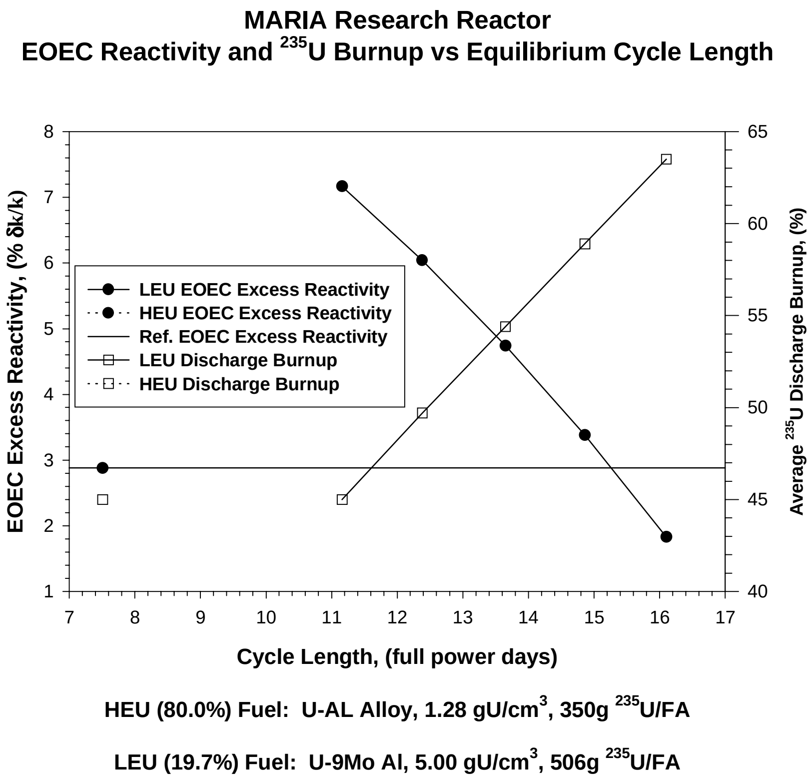


Figure 6

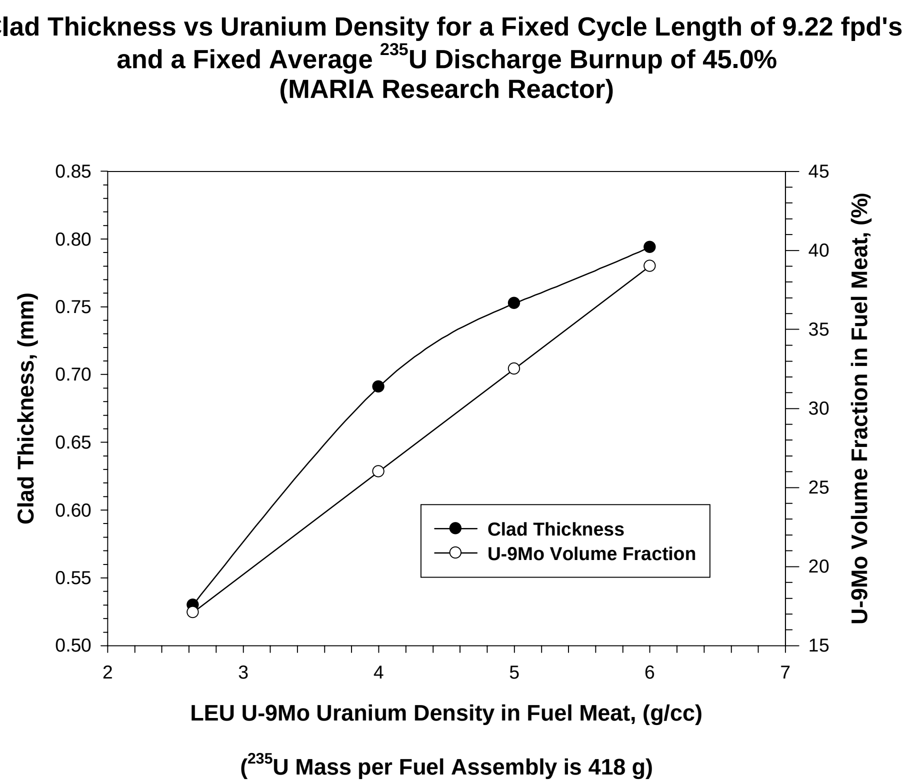


Figure 7

IR-8 Reactor:

Results from REBUS-3 fuel cycle calculations for the LEU U-9Mo (5.00 gU/cm^3) IRT-3M 6-tube fuel assemblies are given in Table 5 and plotted in Fig. 8. Values for the HEU (90%) reference fuel and for the 36%-enriched fuel are reproduced from Ref. 4. For all these cases the coolant channel thickness was 2.05 mm. As before, the BOEC and EOEC excess reactivities have been adjusted for the DIF3D/MCNP reactivity bias calculated for fresh fuel. The plot shows that relative to the reference fuel, use of LEU U-9Mo (5.00 gU/cm^3) fuel would increase the cycle length (for the same EOEC excess reactivity) from 36 to 40 full power days. This results in an 11% decrease in annual fuel consumption. To match the performance of the 36% enriched fuel would require a small increase in the loading of the U-9Mo Al-dispersion fuel (≈5.05 gU/cm^3). Peak thermal neutron fluxes in the F3 location (see Fig. 4) are reduced by about 6% with respect to the HEU (90%) reference fuel.

Table 5. IR-8 Research Reactor Equilibrium Fuel Cycle Comparisons
(Core: 16 IRT-3M 6-Tube Fuel Assemblies, Power = 8.0 MW)

Calc. Type	^{235}U Enr. wt%	Fuel Meat (^{235}U-g/FA)	Cycle length fpd's	Excess Reactivity, % δk/k [a]				Discharge BU		Peak ϕ_{th}-BOEC E+14 n/cm^2-s [b]	
				BOL	Bias	BOEC	EOEC	Ave. %	Peak %	Loc. F3	Loc. B5
MCNP	90.0	UO_2-Al	0.0	20.57	0.0					2.54	1.65
DIF3D	90.0	(264)	0.0	21.61	1.05					2.46	1.53
Rebus3	90.0	(264)	34.0		1.05	8.07	3.71	62.9	72.0	2.46	1.96
Rebus3	90.0	(264)	36.0		1.05	7.08	2.22	66.5	75.6	2.46	1.95
MCNP	36.0	UO_2-Al	0.0	18.91	0.0					2.39	1.47
DIF3D	36.0	(309)	0.0	21.00	2.09					2.33	1.41
Rebus3	36.0	(309)	36.0		2.09	8.15	4.66	56.6	65.3	2.33	1.62
Rebus3	36.0	(309)	38.0		2.09	7.52	3.72	59.6	68.4	2.33	1.64
MCNP	19.75	U-9Mo Al	0.0	16.59	0.0						
DIF3D	19.75	(339)	0.0	18.22	1.63					2.31	1.32
Rebus3	19.75	(339)	36.0		1.63	6.68	3.69	50.8	58.6	2.31	1.45
Rebus3	19.75	(339)	37.0		1.63	6.43	3.33	52.1	60.1	2.31	1.46
Rebus3	19.75	(339)	38.0		1.63	6.17	2.97	53.4	61.4	2.31	1.47
Rebus3	19.75	(339)	39.0		1.63	5.91	2.59	54.7	62.8	2.31	1.48
Rebus3	19.75	(339)	40.0		1.63	5.64	2.20	56.0	64.2	2.31	1.48

[a] Excess reactivities at the beginning of the equilibrium cycle (BOEC) and at the end of the equilibrium cycle (EOEC) have been corrected for the DIF3D/MCNP reactivity bias.
[b] Neutron fluxes are multiplied by k_{eff} to adjust them to the critical core condition. ϕ_{th} is the neutron flux for energies below 0.625 eV.

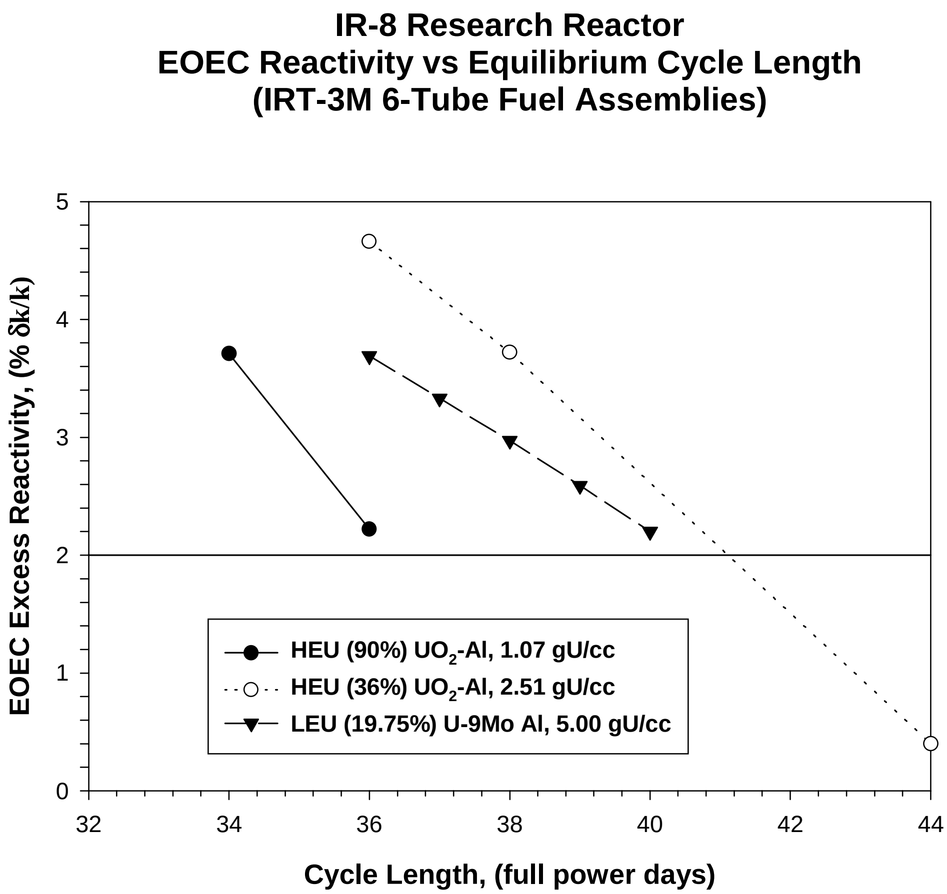


Figure 8

WWR-SM Reactor:

The WWR-SM reactor uses both 6-tube and 8-tube IRT-3M fuel assemblies (see Fig. 3) each with a coolant channel thickness of 2.05 mm. Results from REBUS-3 equilibrium fuel-cycle calculations are summarized in Table 6. For the HEU cases (90% and 36% enriched) data is taken from Ref. 6. At the time of these earlier calculations the reactor used the 90% enriched reference fuel. Conversion to 36%-enriched fuel was completed in 1999. Excess reactivities at BOEC and EOEC have been adjusted for the DIF3D/MCNP bias calculated with fresh fuel in the reactor.

Table 6. WWR-SM Research Reactor Equilibrium Fuel Cycle Comparisons (Core: 20 IRT-3M 6-Tube and 4 IRT-3M 8-Tube Fuel Assemblies, Power = 10.0 MW)

Calc. Type	^{235}U Enr. wt%	Fuel Meat (^{235}U-g/FA)	Cycle length fpd's	Excess Reactivity, % δk/k [a]				Discharge BU 6/8 Tube		Peak ϕ_{th}-BOEC E+14 n/cm^2-s [b]	
				BOL	Bias	BOEC	EOEC	Ave. %	Peak %	Loc. F2	Loc. D5
MCNP	90.0	UO_2-Al	0.0	16.64	0.0						
DIF3D	90.0	(264/ 300)	0.0	15.95	-0.696						
Rebus3	90.0	"	21.0		-0.696	5.96	3.85	43/67	51/77	1.95	3.16
Rebus3	90.0	"	24.0		-0.696	4.44	1.87	49/74	63/84	1.96	3.18
MCNP	36.0	UO_2-Al	0.0	17.01	0.0						
DIF3D	36.0	(309/ 351)	0.0	16.91	-.0924						
Rebus3	36.0	"	21.0		-.0924	6.73	5.13	36/57	43/67	1.85	3.03
Rebus3	36.0	"	24.0		-.0924	5.75	3.88	41/64	49/74	1.85	3.04
MCNP	19.75	U-9Mo Al	0.0	13.45	0.0						
DIF3D	19.75	(339/ 385)	0.0	13.34	-0.107					1.77	2.89
Rebus3	19.75	"	21.0		-0.107	5.14	3.71	32/52	39/61	1.80	2.92
Rebus3	19.75	"	22.0		-0.107	4.87	3.37	34/54	40/63	1.80	2.92
Rebus3	19.75	"	23.0		-0.107	4.59	3.02	35/56	42/66	1.80	2.92
Rebus3	19.75	"	24.0		-0.107	4.31	2.66	37/58	44/68	1.80	2.92
Rebus3	19.75	"	25.0		-0.107	4.02	2.29	38/60	46/70	1.80	2.93
Rebus3	19.75	U-9Mo Al [c]	21.0	13.76	-0.107	5.96	4.62	31/50	37/58	1.77	2.89
Rebus3	19.75	(359/ 408)	24.0		-0.107	5.19	3.66	35/55	41/65	1.78	2.90

[a] Excess reactivities at the beginning of the equilibrium cycle (BOEC) and at the end of the equilibrium cycle (EOEC) have been corrected for the DIF3D/MCNP reactivity bias.
[b] Neutron fluxes are multiplied by k_{eff} to adjust them to the critical core condition. ϕ_{th} is the neutron flux for energies below 0.625 eV.
[c] For these cases the density was increased from 5.00 to 5.30 gU/cm^3.

Figure 9 is a plot of the adjusted EOEC excess reactivities as a function of the equilibrium cycle length. For the reference HEU (90%) IRT-3M fuel assemblies, the 21-day cycle length with 43% average discharge burnup in the 6-tube FA's approximately matches the operating experience of the reactor. Therefore, this value for the EOEC excess reactivity (3.85% δk/k) was used as a basis for comparing the performance of other fuels with this reference fuel even though the reactivity is larger than what is observed experimentally. The burnup model does not include reactivity losses from experiments and additional beryllium poisoning effects, especially in the four central beryllium blocks, that are not already included in the rough estimates for the ^{3}He and ^{6}Li concentrations.

Figure 9 shows that the LEU U-9Mo (5.00 gU/cm^3) Al-dispersion fuel very nearly matches the performance of the HEU (90%) reference fuel for a 21-day cycle length. However, the uranium loading in the U-9Mo fuel would have to be increased to about 5.4 gU/cm^3 to match the EOEC excess reactivity (3.88% δk/k) of the 36% enriched fuel now in use. Relative to the reference case with HEU (90%) fuel, thermal neutron fluxes in locations F2 and D5 (see Fig. 5) will be reduced by about 8%.

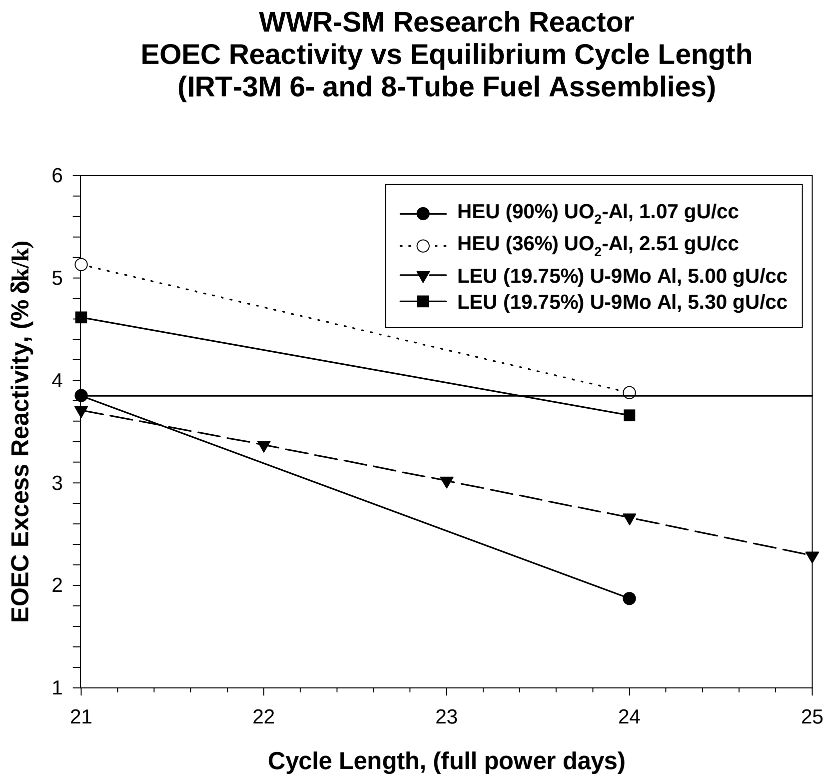


Figure 9

CONCLUSIONS

The performance of LEU U-9Mo (5.00 gU/cm^3, 32.5% U-9Mo by volume) Al-dispersion fuel assemblies relative to HEU (80% or 90%) reference fuel has been evaluated for the MARIA, the IR-8 and the WWR-SM research reactors. In all three cases the fuel tube and coolant channel thicknesses are unaltered although the fuel meat thicknesses are increased. With meat thickness increased from 0.40 to 0.60 mm and clad thickness decreased from 0.80 to 0.70 mm in the MARIA reactor M6 fuel assemblies, this LEU fuel would double the cycle length of the HEU (80%) reference fuel and so reduce fuel consumption by a factor of two. However, the average discharge burnup would be increased from 45% to about 61% and thermal neutron fluxes in experiment locations would be reduced by 7.5-8.0%. For the identical geometry as used in the HEU M6 fuel assemblies and for the same average discharge burnup (45%), a loading in the LEU U-9Mo Al-dispersion fuel meat of about 6.0 gU/cm^3 would be required.

For the IR-8 reactor, use of LEU U-9Mo (5.0 gU/cm^3) in the IRT-3M Al-dispersion fuel assemblies would increase the cycle length from 36 full power days for the HEU (90%) UO_2-Al (1.07 gU/cm^3) reference fuel to 40 full power days. This increase represents about an 11% decrease in annual fuel consumption. The peak discharge burnup of this LEU fuel would be about 64% compared with 76% for the reference case. Thermal neutron fluxes in experiment holes would be lowered by about 6%. To equal the performance of 36% HEU fuel, the uranium density in the U-9Mo Al-dispersion fuel meat would need to be increased from 5.00 gU/cm^3 to about 5.05 gU/cm^3.

Use of LEU U-9Mo (5.0 gU/cm^3) in the 6-tube and 8-tube IRT-3M Al-dispersion fuel assemblies for the WWR-SM reactor would approximately match the performance of the 90% enriched reference fuel. To duplicate the performance of the 36% enriched UO_2-Al fuel assemblies in current use, however, will require a uranium density of about 5.4 gU/cm^3 for the LEU U-9Mo Al-dispersion fuel assemblies.

Relative to the reference fuel, thermal neutron fluxes in experiment locations are expected to decrease by about 8%.

For the three reactors studied, elements with U-9Mo Al-dispersion fuel and a uranium density of 5.00 g/cm^3 (32.5% U-9Mo by volume) would provide the same or better than the HEU (80%-90%) annual fuel consumption rate without changing the thickness of the current elements. Thermal neutron fluxes in experiment holes would be reduced by 6-8%. Full-size U-9Mo M6 and IRT-3M Al-dispersion fuel assemblies need to be fabricated, irradiated and examined before they cab be qualified for research reactor use.

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